Tennessee Valley Authority, Post Office Box 2000, Soddy Daisy, Tennessee 37384-2000

November 10, 2011

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Unit 1
Facility Operating License No. DPR-77
NRC Docket Nos. 50-327

Subject: **Licensee Event Report 327/2011-003, "Unit 1 Reactor Trip As a Result of Turbine Control Card Failure," Supplement 1**

Reference: Letter from TVA to NRC, "Revised Submittal Schedule for Supplemental Report for License Event Report 327/2011-003, 'Unit 1 Reactor Trip As a Result of Turbine Control Card Failure'", dated September 30, 2011

As indicated in the reference letter, the Tennessee Valley Authority (TVA) has completed the evaluation of the licensee event report (LER) reported under 327/2011-003. The enclosed LER has been revised with supplemental information concerning an automatic reactor trip and automatic engineered safety feature actuation of auxiliary feedwater following the failure of a turbine control analog electro-hydraulic signal conditioning card. On August 24, 2011, the TVA submitted Revision 0 of the enclosed LER. At that time, TVA was completing the root cause evaluation for the event. TVA has completed the root cause evaluation and is providing this LER supplement. The revisions are annotated by a vertical bar to the right of the text.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A), a condition that resulted in automatic actuation of the reactor protection system.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact G. M. Cook, Sequoyah Site Licensing Manager, at (423) 843-7170.

IE22
NRR

Respectfully,

John T. Carlin
Site Vice President
Sequoyah Nuclear Plant

Enclosure: Licensee Event Report - Unit 1 Reactor Trip As a Result of Turbine Control Card Failure - Supplement 1

cc: NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/20
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resources@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Sequoyah Nuclear Plant Unit 1	05000327	1 OF 6

4. TITLE:
Unit 1 Reactor Trip As a Result of Turbine Control Card Failure - Supplement 1

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	26	2011	2011 -	003 -	01	11	10	2011	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
SQN - Donald Sutton	(423) 843-6539

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JJ	ECBD	W120	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On June 26, 2011, at 1615 Daylight Saving Time (DST), Sequoyah Nuclear Plant (SQN) Unit 1 received an automatic reactor trip as a result of the failure of a turbine control analog electro-hydraulic (AEH) signal conditioning card. The card failure created an indication that the main turbine had tripped (auto stop latch signal) resulting in the governor and throttle valves closing. When the throttle valves closed, an automatic reactor trip was received from the turbine trip at greater than 50 percent rated thermal power. An immediate corrective action was to replace the failed AEH signal conditioning card. The root cause of this event was determined to be a failure to eliminate and mitigate single point vulnerabilities in AEH signal conditioning printed circuit boards.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Sequoyah Nuclear Plant Unit 1	05000327	2011	-- 003 --	01	2 OF 6

NARRATIVE

I. PLANT CONDITION(S)

At the time of the event, Sequoyah Nuclear Plant (SQN) Unit 1 was operating at approximately 100 percent rated thermal power with the turbine control system in normal operation.

II. DESCRIPTION OF EVENT

A. Event:

On June 26, 2011, at 1615 Daylight Saving Time (DST), SQN Unit 1 automatically tripped as a result of the failure of a turbine control analog electro-hydraulic (AEH) signal conditioning card [EIIS Code JJ]. Failure of the card created an indication that the turbine had tripped and resulted in the governor and throttle valves [EIIS Code TA] closing. Prior to the reactor trip, reactor power was at approximately 100 percent. Following the reactor trip, the steam dump [EIIS Code SG] system functioned initially as expected (all valves opened). Subsequently, the steam dump system was manually turned off because three of the valves did not close when expected. As a consequence, decay heat removal was via the steam generator atmospheric relief valves.

The Tennessee Valley Authority is submitting this report in accordance with 10 CFR 50.73 (a)(2)(iv)(A), a condition that resulted in automatic actuation of the reactor protection system.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

Date	Description
June 26, 2011 at 1615:26 DST	Turbine governor valves start closing due to the signal conditioning card failure.
June 26, 2011 at 1615:29 DST	Control rods begin stepping in as a result of the turbine load reduction.
June 26, 2011 at 1615:36 EDT	Operators misdiagnose reason for control rod insertion and incorrectly place control rods in manual.
June 26, 2011 at 1615:47 DST	All four turbine throttle valves closed. Automatic reactor trip occurs due to turbine trip above 50 percent rated thermal power. Operations entered Emergency Procedure E-0 "Reactor Trip or Safety Injection".

NARRATIVE

June 26, 2011 at 1625:50 DST	Three steam dump valves did not close as expected following the trip. Operators turn steam dump control off in accordance with Emergency Subprocedure ES-0.1 "Reactor Trip Response," a subprocedure of procedure E-0. Decay heat removal via the steam generator atmospheric relief valves was used in accordance with Emergency Subprocedure ES-0.1.

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

Control room alarms alert operators to the start of the event.

F. Operator Actions:

At the time of the turbine load rejection, control bank D rods automatically inserted 10 steps over 7 seconds from an initial rod height of 220 steps withdrawn. The reactor operator misdiagnosed the reason for the insertion and incorrectly placed the control rods in manual; the reactor tripped approximately 12 seconds later with all rods fully inserting. Had the rods remained in automatic, rod height of control bank D would have been approximately 198 steps withdrawn at the time of the reactor trip. The difference in reactivity worth from the actual versus projected rod height of control bank D is not significant and is bounded by the accident analysis that assumes the control rod of highest worth remains fully withdrawn. The operator's actions of placing the rods in manual was not in compliance with plant procedures and did not meet expectations set by Operations Management or Training. Once the reactor tripped, Operations entered emergency operations procedure E-0 "Reactor Trip or Safety Injection," as required by procedure. The steam dump system initially functioned as expected (all valves opened); afterwards, Operations manually turned off the steam dump system because three of the valves did not close when expected.

G. Safety System Responses:

The plant responded as expected for the conditions of the reactor trip.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Sequoyah Nuclear Plant Unit 1	05000327	2011	-- 003	– 01	4 OF 6

NARRATIVE

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the reactor trip was a turbine trip above 50 percent rated thermal power due to a failed turbine control AEH card.

B. Root Cause:

The root cause of this event was determined to be a failure to eliminate and mitigate single point vulnerabilities in AEH signal conditioning printed circuit boards.

C. Contributing Factor:

A contributing factor was the lack of preventative maintenance or testing to mitigate or identify AEH printed circuit board (PCB) degradation.

IV. ANALYSIS OF THE EVENT

SQN Unit 1 was operating in Mode 1 at approximately 100 percent rated thermal power. At 1615 DST, the reactor automatically tripped following a turbine trip from greater than 50 percent rated thermal power (P-9 interlock). Prior to the event, reactor coolant system (RCS) pressure was approximately 2234 pounds per square inch gauge (psig). Following the turbine control card failure, the turbine valves closed and the turbine load reduction caused a rise in RCS temperature. This caused RCS volume to increase with a corresponding increase in RCS pressure. RCS pressure increased above 2335 psig, the setpoint of the pressurizer power operated relief valves (PORVs), before the reactor trip. Both PORVs opened briefly (approximately 3 seconds); approximately 7 seconds later, one PORV re-opened. Both PORVs reclosed at the proper pressure. Because of the turbine load reduction, RCS average temperature increased to approximately 586 degrees Fahrenheit, which is 3 degrees Fahrenheit above the Technical Specification (TS) 3.2.5 limit of 583 degrees Fahrenheit. The TS 3.2.5, Departure from Nucleate Boiling (DNB) Parameters, action statement requires that the parameter be restored to a value within its limits within 2 hours or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 4 hours. The TS 3.2.5 limit for average temperature was exceeded for a period of approximately 7 seconds which is well within the TS 3.2.5 action statement requirement. Following the reactor trip, average temperature rapidly decreased as a result of loss of nuclear heat generation to approximately 545 degrees Fahrenheit, then increased to its no-load value of 547 degrees Fahrenheit. Following the trip, all safety related equipment operated as designed. The auxiliary feedwater system automatically actuated as expected on loss of the main feedwater pumps. The main feedwater pumps were available for recovery using approved plant procedures following the scram. Initially, the steam dump system functioned as expected (all valves opened). Subsequently, the steam dump system was

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Sequoyah Nuclear Plant Unit 1	05000327	2011	-- 003	-- 01	5 OF 6

NARRATIVE

manually turned off because three of the valves did not close when expected. As a consequence, decay heat removal was provided via the atmospheric relief valves.

V. ASSESSMENT OF SAFETY CONSEQUENCES

As discussed in the above "Analysis of The Event," following the trip, all safety related equipment operated as designed, the auxiliary feedwater system actuated as expected and decay heat removal was provided using the atmospheric relief valves. In addition, the DNB parameter for average temperature momentarily increased to above the TS 3.2.5 limit to 586 degrees Fahrenheit. However, the parameter was restored to within the TS 3.2.5 limit well within the allowance of the associated action statement and all other DNB parameters remained within limits during this event. As a result, this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

The immediate corrective action was to replace the failed AEH signal conditioning card.

B. Corrective Actions to Prevent Recurrence:

The corrective actions are being managed through the SQN Corrective Action Program.

A corrective action to prevent recurrence is to implement a PCB lifecycle management program to provide preventative maintenance combined with monitoring and planning tasks to increase PCB reliability. In addition, a single point vulnerability (SPV) program will be implemented to identify, provide guidance, and mitigate SPVs. Additional actions are being taken and are captured within the Corrective Action Program under problem evaluation report number 393838.

VII. ADDITIONAL INFORMATION

A. Failed Components:

The failed component was a Westinghouse AEH signal conditioning card.

B. Previous LERs on Similar Events:

A review of previous reportable events for the past three years did not identify any previous similar events.

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REV NO.	
Sequoyah Nuclear Plant Unit 1	05000327	2011	-- 003	-- 01	6 OF 6

NARRATIVE

C. Additional Information:

None.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with 10 CFR 50.73(a)(2)(v).

E. Unplanned Scram with Complications:

This event did not result in an unplanned scram with complications. The main feedwater pumps were available for recovery using approved plant procedures following the scram. Although the steam dump system valves opened as expected, the steam dump system was later turned off because three of the valves did not close when expected. The atmospheric relief valves were available and were subsequently used for decay heat removal.

VIII. COMMITMENTS

None.